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Comparison of Low-Level Waste Disposal Programs of DOE and Selected International Countries

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ABSTRACT

This document is a result of the Secretary of Energy's response to Defense Nuclear Facilities Safety Board Recommendation 94-2. The Secretary stated that the U.S. Department of Energy (DOE) would "address such issues as...the need for additional requirements, standards, and guidance on low-level radioactive waste management." The authors gathered information and compared the disposal programs used by the U.S. DOE, France, Sweden, Canada, and the United Kingdom to dispose of low-level radioactive waste. The study identified many similarities in practices but also identified some differences in disposal practices and national policies.

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Comparison of Low-Level Waste Disposal Programs of DOE and Selected International Countries

1. INTRODUCTION

The purpose of this report is to examine and compare the approaches and practices of selected countries for disposal of low-level radioactive waste (LLW) with those of the United States (U.S.) Department of Energy (DOE). The report addresses the programs for disposing of wastes into engineered LLW disposal facilities and is not intended to address in-situ options and practices associated with environmental restoration activities or the management of mill tailings and mixed LLW. The countries chosen for comparison are France, Sweden, Canada, and the United Kingdom. The countries were selected as typical examples of the LLW programs which have evolved under differing technical constraints, regulatory requirements, and political/social systems. France was the first country to demonstrate use of engineered structure-type disposal facilities. The UK has been actively disposing of LLW since 1959. Sweden has been disposing of LLW since 1983 in an intermediate-depth disposal facility rather than a near-surface disposal[a] facility. To date, Canada has been storing its LLW but will soon begin operation of Canada's first demonstration LLW disposal facility.

The percentage of LLW resulting from electrical power reactor operations is 90, 77, 50, 43, and 0 percent for France, the UK, Sweden, Canada, and the DOE respectively. The remaining wastes of the various countries come from traditional activities such as defense activities, medical applications, research and development, and industrial applications. U.S. electrical power reactor operations are controlled by the U.S. Nuclear Regulatory Commission (NRC), which is not included in this study. The DOE does, however, have wastes resulting from specialty reactor operations, which include naval vessel propulsion, test reactors, and production of weapons materials. Other typical DOE operations include high enrichment of uranium; weapons research, development and fabrication; nuclear fuel reprocessing; and other forms of basic nuclear research and development.[1] A larger amount of DOE wastes are generated from activities other than electrical power reactors, than from the programs of other countries studied. However, as stated later in this report, the waste classification systems used by the DOE and the countries studied are quite similar, so it is appropriate to compare disposal activities to identify different practices of other nations. For the programs studied, wastes falling into the category of LLW are deemed as being suitable for near-surface disposal. These wastes have the general characteristics that they do not contain large concentrations of long-lived radionuclides or beta/gamma activities which generate amounts of heat significant enough to affect the disposal facility design.

Table 1 shows the nuclear fuel cycle activities performed by each of the nuclear programs included in this study. The fuel cycle activities shown for DOE are associated with the fabrication and reprocessing of defense-related reactor materials. Sweden has the least number of fuel cycle-related nuclear activities. The waste-types and the average isotopic make-up of the wastes may vary somewhat from one program to another due to differences in origins of the wastes. These facts may affect site-specific disposal facility performance assessment, waste classification considerations, waste acceptance criteria, and waste handling considerations.

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a. As defined in 10 CFR 61.7(a), "near-surface disposal" involves disposal in the uppermost portion of the earth, approximately 30 meters. Near-surface disposal includes disposal in engineered facilities that may be built totally or partially above-grade provided that such facilities have protective earthen covers.

This report should be used as an overview document to identify differing and similar practices of waste disposal programs. Summary considerations associated with LLW policy, disposal facility design, performance assessment, and operation are addressed. It should be noted that subtle and important differences between published information and actual practice may exist. Some of these important differences have been revealed during the preparation and review of this report and are presented. Further detailed investigation of national practices may be necessary to fully understand the advantages and disadvantages of other nation's waste management programs.

The appendix to this document contains a description of the LLW disposal programs and additional details for each of four countries and the U.S. DOE.

Much of the information used to assemble this report was identified using librarian-assisted searches of commercially-kept literature databases and from literature obtained from personnel who have toured or have employment ties with the foreign facilities. Time and resources did not allow visits to the various countries or personal contact with individuals at the various disposal facilities.

The authors have avoided drawing conclusions or making judgments concerning differences in approaches and practices of the various nations. In addition, no recommendations on certain international aspects that could be considered for implementation in the DOE program have been offered.

Table 1. Nuclear activities performed by the nations under study.

Program	Uranium production	Uranium ore conversion	Uranium enrichment	Fuel fabrication	Fuel reprocessing
U.S. DOE	*	*	X	X	X
United Kingdom		X	X	X	X
France	X	X	X	X	X
Sweden				X	
Canada	X	X		X	

X Indicates that the column topic is performed within the indicated program.

* These activities are performed by the U.S. commercial industry, but not by the DOE.

2. REGULATORY APPROACH AND APPROVAL REQUIREMENTS

In all cases studied, the federal government of each country establishes the policies and regulations for the design, construction, and operation of waste management facilities. Table 2 lists the federal agencies of each country that establish policies for waste disposal and that give formal approval to construct and operate a LLW disposal facility in each of the programs included in this study. All programs studied, except the U.S. DOE and Canadian programs, have a federal agency that is administratively independent of the waste disposal activities to approve disposal facility activities and to regulate and oversee the operation of the facilities. A recent federal advisory committee for DOE

Table 2. LLW disposal facility approval requirements and public involvement.

Program	Policies established by	Final approval agency	Public involvement	Regulator	Operator
U.S. DOE	Federal Government (Department of Energy)	DOE-HQ	Public hearing associated with the programmatic or site-wide Environmental Impact Statement.	Federal Government Agency (DOE-HQ)	Government Agency (DOE) Subcontractors
United Kingdom	Federal Government (Department of the Environment)	Health and Safety Executive	Local planning authority (county or borough) must review the proposal and address written public comments. If it refuses to approve the proposal, the applicant may appeal to the Secretary of State who may overrule the local authority after a public hearing.	Federal Government Agency (Health and Safety Executive)	Government-Owned Corporation (NIREX)
France	Federal Government (Atomic Energy Commission)	French Prime Minister	A "Public Inquiry" is held, making all information available to the public. Written public comments are evaluated by an "Inquiry Commission" appointed by the local administrative court. After a report is completed by the Commission and circulated to the public, communities within a 5-km radius of the project are allowed to vote for or against the project. A negative vote can be overruled by the national Parliament.	Federal Government Agency (Ministry of Industry)	Government Agency (ANDRA) Subcontractors
Sweden	Federal Government (Ministry of Environment and Energy)	Nuclear Power Inspectorate	The local municipal council, as well as the Federal Government licensing agency (Nuclear Power Inspectorate), must approve the licensing of the disposal facility.	Federal Government Agencies (Nuclear Power Inspectorate and National Institute of Radiation Protection)	Private Company (Nuclear Fuel and Waste Management Company)
Canada	Federal Government (Atomic Energy Control Board)	Atomic Energy Control Board	Most new nuclear facilities in Canada are referred to the Federal Minister of Environment for a formal public review by an independent panel, with full opportunity for public hearings, and funding for intervenors.	Federal Government Agency (Atomic Energy Control Bureau)	Generator [Government (Atomic Energy of Canada Limited) owns 90% of existing waste]

investigated the DOE approval policy and has recommended that the DOE be subject to control by an independent regulatory agency.[2]

Political and social considerations are recognized by most countries as crucial to gaining acceptance and approval of a LLW disposal facility. All programs studied allow the public to have input during the approval process for the facility. Table 2 describes how each country involves the general and local public in the approval process for LLW disposal facilities.

France is the only country studied that requires a public vote before licensing a new nuclear waste disposal facility. A negative vote can only be overturned by the national Parliament. The UK and Sweden require approval by a local representative council before a disposal project can be authorized. In the case of a negative vote, the Secretary of State of the UK may overrule the local authority after a public hearing. No overrule of the local municipal council is currently allowed by Sweden. Both the U.S. DOE and Canada hold public meetings to solicit public comment on the facility, but no formal local or public approval is required for those governments to approve the facility and begin construction and operation of a LLW disposal facility.

3. WASTE CLASSIFICATION

An initial step in comparing and discussing the LLW disposal approaches of various programs is to determine if the programs use similar criteria to identify LLW. Table 3 identifies the basic criteria used by each program to define LLW. The table also shows the classification criteria recommended as typical characteristics of waste classes by the International Atomic Energy Agency (IAEA)[3]. (Note: Use of IAEA criteria is not mandatory for any nation). As noted in the table, the recommended IAEA criteria have been adopted for use by Sweden.

Although there are differences in classification of LLW, all programs recognize that LLW disposed in near-surface disposal facilities should not contain significant quantities of the longer half-lived radioactive isotopes. All programs plan deep geologic disposal facilities for spent fuel and long-lived wastes (excluding mill-tailing type wastes). Since the alpha-emitting radionuclides typically have long half-lives, many programs have chosen to limit their concentrations in waste classified as LLW, as shown in Table 3. The alpha-emitting isotopes with half-lives greater than 20 years are listed in Table 4. The following differences are noted regarding acceptable maximum alpha-emitter concentrations:

> The UK, France, and Sweden all limit alpha-emitting radionuclide concentrations to approximately 100 nCi/g. France implements this requirement differently than other programs in that the long-lived alpha-emitter limit is as calculated 300 years after time of acceptance. France also allows up to 500 nCi/g per container, on a limited exception basis.
>
> DOE Order 5820.2A limits only alpha-emitting transuranium elements (having half-lives greater than 20 years) in LLW to concentrations less than 100 nCi/g. Alpha-emitters which are not transuranium isotopes (and have half-lives greater than 20 years) are identified in Table 4. These lighter alpha-emitter elements include uranium, thorium, and radium. DOE Order 5820.2A authorizes Heads of Field Elements to specify that other alpha-contaminated wastes, peculiar to a specific site, must be managed as transuranic wastes.
>
> Canada acknowledges that the LLW must be isolated for a period up to 500 years but has not yet assigned quantitative values on the long-lived elements. This undefined classification is meant to include those wastes that will be acceptable for near-surface disposal in the proposed Intrusion Resistant Underground Structure (IRUS) discussed later. Technical

Table 3. Comparison of low-level waste classification systems.

Program	LLW classification parameters[a]	Alpha limits	Disposal
U.S. DOE	Low-level waste - Not transuranic (<100 nCi/g transuranium alpha-emitting radionuclides with half-lives greater than 20 years); not high-level waste; not spent fuel; and not mill tailings.	<100 nCi/g transuranium alpha-emitters (half-life >20 years), Field Elements can further limit alpha activity.	Near-surface disposal.
United Kingdom	Low-level waste - <4 GBq/MT (108 nCi/g) alpha-emitting and <12 GBq/MT (324 nCi/g) beta-gamma.	<108 nCi/g alpha-emitters	Presently near-surface disposal; with deep geologic disposal planned in the future.
	Very low-level waste - <0.0004 GBq/MT (10.8 pCi/g) alpha-emitting and <0.02 GBq/MT (540 pCi/g) beta-gamma.	<10 pCi/g alpha-emitters	Land-disposed with ordinary domestic wastes.
France	Category A - Long-lived alpha-emitters with half-lives >31 years (as calculated 300 years after time of acceptance) <100 nCi/g (0.1 Ci/MT) per container; <10 nCi/g (0.01 Ci/MT) average per container; 100-500 nCi/g per container are accepted on a limited exception basis.	<100 nCi/g alpha-emitters per container <10 nCi/g alpha-emitters average	Near-surface disposal.
Sweden	IAEA recommended waste classifications are used. (See IAEA)	<108 nCi/g alpha-emitters per container <11 nCi/g alpha-emitters average.	Crystalline rock, 60 meters under the Baltic Sea floor.
Canada	Low-level waste - Not high-level waste, spent fuel, or mill tailings. Wastes that require isolation for up to 500 years. (Quantitative parameters for this requirement are in development and not yet well defined - see Section 3 discussion).	Not specified. TBD by Safety Analysis Report.	Near-surface disposal.
	Releasable waste - Decided on a case-by-case basis based on a de minimis dose to individuals of 0.05 mSv/yr (5 mrem/yr).	Based on calculation.	Near-surface disposal.
IAEA	Short-lived low-level and intermediate level wastes[b] - alpha-emitting radionuclides <108 nCi/g (4,000 Bq/g) per container; <11 nCi/g (400 Bq/g) average in the disposal facility; thermal power <2 kW/m^3.	<108 nCi/g alpha-emitters per container. <10.8 nCi/g alpha-emitters average.	Near-surface or geological disposal facility.
	Exempt wastes - Activity levels at or below clearance levels, which are based on an annual dose to the public <0.01 mSv (1 mrem). Suggested radionuclide-specific clearance levels are proposed and are currently issued for comment (Reference 4).	Calculated based on allowable exposure.	No radiological restrictions.

a Each disposal site may further limit the LLW characteristics allowable for disposal in their waste acceptance criteria, based on the facility performance assessment documentation.

b Under the proposed IAEA classification system, low-level waste and intermediate level wastes have the same radioactivity concentration limits but are distinguished by the shielding requirements typically imposed when handling the wastes. Intermediate level waste typically requires shielding during handling activities, whereas low-level waste does not. A contact dose rate of 2 mSv/hr (200 mrem/hr) is typically used to as the quantitative limit to separate the two categories of waste.

Table 4. Alpha-emitting radionuclides with half-lives greater than 20 years.

Radionuclide	Isotope	Transuranium	Half life (yrs)
Americium	Am-241	Yes	4.32e+02
Americium	Am-242m	Yes	1.52e+02
Americium	Am-243	Yes	7.38e+03
Californium	Cf-249	Yes	3.51e+02
Californium	Cf-251	Yes	9.00e+02
Curium	Cm-243	Yes	2.85e+01
Curium	Cm-245	Yes	8.50e+03
Curium	Cm-246	Yes	4.75e+03
Curium	Cm-247	Yes	1.56e+07
Curium	Cm-248	Yes	3.39e+05
Neptunium	Np-237	Yes	2.14e+06
Plutonium	Pu-238	Yes	8.78e+01
Plutonium	Pu-239	Yes	2.41e+04
Plutonium	Pu-240	Yes	6.54e+03
Plutonium	Pu-242	Yes	3.76e+05
Plutonium	Pu-244	Yes	8.26e+07
Neodymium	Nd-144	No	2.10e+15
Protactinium	Pa-231	No	3.28e+04
Radium	Ra-226	No	1.60e+03
Samarium	Sm-147	No	6.90e+09
Samarium	Sm-148	No	7.00e+15
Thorium	Th-229	No	7.34e+03
Thorium	Th-230	No	7.70e+04
Thorium	Th-232	No	1.40e+10
Uranium	U-232	No	7.20e+01
Uranium	U-233	No	1.59e+05
Uranium	U-234	No	2.45e+05
Uranium	U-235	No	7.04e+08
Uranium	U-236	No	3.42e+06
Uranium	U-238	No	4.47e+09

issues, including defining more detailed waste classification limits, are currently being reviewed and resolved by the Canadian authorities. More definitive limits will be established in the facility safety documentation.

Comparison of the above quantitative values should not be taken strictly at face value since some programs implement the limits more conservatively than others. The most notable differences in practices are best illustrated by comparing certain practices of France and DOE. Requirements found in French LLW disposal facility waste acceptance criteria allow the following practices in meeting the concentration limit:[5]

> The mass of the waste container and any solidification agent is included in calculating alpha concentrations.
>
> Alpha concentrations are determined as projected 300 years after the time of acceptance, not at the time of disposal.
>
> Higher alpha concentrations, up to 500 nCi/g, are allowed on a limited basis by special permission by ANDRA and the safety authorities.

In comparison, the U.S. DOE does the following:

> The mass of the waste matrix (excluding the disposal container) is used to calculate alpha-emitter concentrations for purposes of determining if the waste is LLW or transuranic waste. Varying practices are prescribed in site-specific facility waste acceptance criteria concerning how the weight of the waste container is used in calculating radionuclide concentration for purposes of disposal.
>
> Alpha concentrations at the time of initial characterization are used, rather than a projected value to a date in the future.
>
> Exceptions are not allowed to the maximum concentration limit for alpha-emitters in LLW. Those wastes which exceed this limit are classified as transuranic waste.

These practices can make an appreciable difference in allowable waste that may be called LLW. For example, allowable concentrations for Pu-238, which has an 86 year half-life, could be approximately a factor of 10 higher under the French compared to DOE requirement for reporting radionuclide concentrations.

Another difference in classification of LLW is the criteria used by the various programs and the IAEA to generically limit <u>beta-gamma</u> activity in the waste. The following differences can be noted from Table 3:

> The DOE, France, and Canada define this beta-gamma limit by exclusion. That is, if a waste is radioactive, but it is not high-level waste, transuranic or greater-than-Class C waste (applicable to the U.S. DOE only), uranium mill tailings, or spent fuel, then it is classified as LLW.
>
> The UK specifies quantitative activity concentration limits for beta-gamma activity in its LLW classification scheme.

Sweden uses the IAEA system, which defines a maximum heat generation criteria (2 kW/m^3) for the waste, which is related to the beta-gamma activity.

The UK and Sweden have defined generic quantitative limits for beta-gamma activity in their LLW as part of their waste classification schemes. Waste disposal facilities sometimes further limit beta-gamma emitters, on an isotope specific basis, in their waste acceptance criteria, based on the analytical results of the site-specific performance assessment. The IAEA classification guideline, which is not mandatory for any country to use, suggests no quantitative limits for beta-gamma activity but does note that limits on some radionuclides may be imposed on a disposal facility, site-specific basis. As already noted, the heat generation criteria suggested by the guideline is related to the beta-gamma activity. Variations in allowable beta-gamma activity may affect the shorter-term waste handling considerations since these isotopes are largely the source of the penetrating radiation that drives remote handling and shielding considerations while handling the waste containers during storage and disposal operations. Long-lived beta-gamma emitters, such as C-14, I-129, Tc-99, Ni-59, and Nb-94, are also important considerations in the long-term institutional control considerations of the disposal facility.

Compared to the other nations studied, Canada s LLW waste program is still in its infancy. Up to now, Canada has been storing all of its radioactive wastes and is now in the process of developing its first near-surface LLW disposal demonstration facility. Canada's disposal efforts are still in the demonstration phases and acceptable waste loadings are established as a part of the facility final safety assessment report.[6] This prototype demonstration facility is scheduled to begin operation in the 1998-99 time frame. The country will likely establish quantitative disposal limits, applicable to all LLW, as the disposal program matures.

Table 3 also identifies that the UK, Sweden, and Canada have defined criteria to allow non-licensed disposal of radioactive wastes that have sufficiently low-levels of radioactivity.[7,8] These countries have defined that these "exempted" or "very low-level wastes" have limited health risk to the public and can be disposed with little or no radiological restrictions. Britain designates very low-level wastes as having an alpha content below 0.0004 GBq/MT (10 pCi/g) and a beta/gamma content of less than 0.02 GBq/MT (540 pCi/g).[7] Canada approves exempted wastes on a case-by-case basis using a maximum allowable de minimus radiation dose rate to individuals of 0.05 mSv/yr (5 mrem/yr) and provided that the radiological impact will be localized and the potential for exposures to large populations is small.[9] Sweden subscribes to the proposed IAEA exempt waste criteria which defines maximum radionuclide activity clearance levels which are based on a dose rate to members of the public of less than 0.01 mSv/yr (1 mrem/yr).[3,4]

France has not established criteria for exempt wastes. DOE does not have generic release criteria for wastes with volume contamination (such as activated material or smelted contaminated metals), but does allow release of such materials if criteria and survey technics are approved by EH-1.[10,11] Before this released material can be disposed in a DOE or non-DOE landfill, it must meet the acceptance criteria of that facility.

4. DISPOSAL FACILITIES DESIGN

4.1 Design Considerations

All programs studied recognize that the basic objective of the siting process is to select a suitable site for disposal and to demonstrate that the site has characteristics that, when combined with the facility design and waste package, provide adequate isolation of radionuclides from the biosphere for desired periods of time. All countries acknowledge that successful site selection involves many factors, not all of which are technical. Public opinion and receptiveness to the proposed site is a key factor that has, at times, resulted in the abandonment of technically acceptable disposal sites.[12,13] Proximity to large populations and to facilities generating LLW are other key factors. Climate and surface hydrology are also primary considerations in site selection since water is a primary radionuclide transport mechanism. Canada utilizes a voluntary siting process to minimize the public outcry upon site selection. This is the result of opposition encountered when siting was done entirely on a technical basis.

DOE has developed each of its disposal sites on a facility-specific basis as done by individual countries; however, all DOE sites use common performance objectives established in DOE Order 5820.2A. DOE disposal facility locations are constrained to the boundaries of the DOE reservations. Although this is a constraint, it should be recognized that many of the sites were originally chosen with emphasis on favorable characteristics for nuclear activities.

Excluding the disposal program of the U.S. DOE, each national disposal program included in this study currently has only one major LLW disposal facility. For this reason, sites have been chosen and developed on a case-by-case basis by each country, with the common design goal of avoiding corrective actions during the facility operation period and after closure of the facilities. All countries except Sweden currently use near-surface disposal facilities for LLW. Sweden uses intermediate-depth disposal for its LLW, primarily due to the imposition of a requirement that post-closure performance should not be dependent on control or corrective actions.[14] Institutional control requirements for this facility are presented in Section 5.

The UK is currently planning to use deep geologic disposal for LLW when the existing near-surface disposal facility capacity is reached. This decision dates to 1987 when a House of Commons Select Committee recommended that all intermediate level waste (ILW) should be disposed in a deep geologic repository and that the same facility should be extended to take LLW also.[15,16] The Committee believed that putting all ILW in a deep repository would result in a gain in public acceptability. The Secretary of State was advised that a near-surface disposal facility for LLW alone would be uneconomic. The recommendation was accepted, while acknowledging that there is no technical requirement to dispose of short-lived wastes in a deep geologic repository.[13]

Approval of a LLW disposal facility, in any country, involves proving that the disposal system will perform acceptably. Various barriers are used to assure that radionuclide migration is controlled to acceptable levels. These are:

Site geology

Waste form

Engineered structures (e.g., vaults and tumuli)

Engineered surface barriers (e.g., surface caps).

Many factors such as climate, geologic makeup, depth to the water table, and surface water conditions are used in evaluating the need for engineered structure, surface barrier, and waste form requirements. Demographic, economic, socio-political, and institutional factors also play a significant role in defining acceptable disposal solutions in the more populated areas found in Europe and in some areas of the United States.

In the early years of LLW disposal, the site geology was generally regarded as the primary barrier for both short- and long-term facility performance. Gradually, as some disposal sites have experienced evidence of radionuclide migration through the geologic barriers, more emphasis has been placed by the European and some DOE facilities (Savannah River and Oak Ridge) on waste form and engineered structures and barriers as the primary migration deterrents during the earlier phases of the facility existence. Geological barriers assume a more important role as the engineered structures and barriers degrade with time and when human maintenance is no longer provided to inspect and care for barriers such as drainage collection systems. The long-term performance of the facility must rely on the site geology as a primary migration barrier.

Water is normally the primary vehicle supporting radionuclide migration at a LLW disposal site. Table 5 shows the average precipitation for the LLW disposal sites included in this study. As can be seen from the table, the humid disposal sites are those utilizing engineered structures (e.g., vaults and tumuli) in their facility designs. This is primarily due to the increased amounts of water available as a radionuclide migration transport medium. The arid DOE sites do not utilize engineered structures and instead rely on surface barriers (such as covers and caps) as the primary engineered deterrents to keep precipitation from reaching the waste.

Table 5. Climate and precipitation at LLW disposal facilities.

Program	Disposal Facility	Engineered Structure Feature	Average Annual Precipitation (inches)	Climate
United Kingdom	Drigg Site	Concrete Vault	40-42	Humid
France	Centre de l'Aube	Concrete Vault	27.6-33.5	Humid
Sweden	Swedish Final Repository (SFR)	Rock Cavern	18-20	Humid
Canada	Intrusion Resistant Underground Structure (IRUS)	Concrete Vault	20-30	Humid
U.S. DOE	Hanford	None	6.3	Arid
	Idaho National Engineering Laboratory	None	8.7	Arid
	Nevada Test Site	None	4.9	Arid
	Los Alamos National Laboratory	None	13.2	Arid
	Oak Ridge Reservation	Concrete Tumulus	54	Humid
	Savannah River Site	Concrete Vault	48.8	Humid

Each country s program and each facility approaches the design of its disposal system differently, with different emphases placed on site geology, waste form, and engineered structures and surface barriers. The facility technical evaluation (performance assessment) is the basis by which each facility justifies that its system design will prevent radionuclide migration sufficiently to prevent public radiation exposure beyond allowable limits.

4.2 Disposal Practices

Table 6 lists the current LLW disposal site designs being used by the programs included in this study. Designs include near-surface disposal with engineered structures and/or surface barriers, intermediate-depth geologic disposal, and deep geologic disposal. Although none of the countries is currently using deep geologic disposal for LLW, the UK plans to use it in the future when existing near-surface facility capacity is exhausted. The rationale for this decision was discussed earlier in Section 4.1.

The DOE is the only entity studied that currently has multiple LLW disposal sites. Canada may have multiple sites at some future date since its waste generators are responsible for disposal of their own wastes. All other foreign countries studied currently have no intention of developing multiple sites.

As already noted, those DOE facilities located in the more humid areas of the United States (Savannah River and Oak Ridge) use engineered structures (such as vaults and tumuli), as well as surface barriers, to assure that the facility will meet the performance objectives established in DOE Order 5820.2A. The more arid DOE disposal sites (Nevada, Los Alamos, Hanford, and Idaho) do not use engineered structures in their facility design, but do use surface barriers such as water-repellent layers (clay, asphalt, concrete), capillary barriers (hydraulic breaks), and rock layers (riprap, gravel) to minimize water contact with the wastes. Fewer surface barriers are sometimes used at the same DOE disposal site for low activity wastes than for high activity waste due to the limited radionuclide migration potential of the lower activity waste.

Sweden has elected to use intermediate-depth geologic disposal for its LLW. This crystalline-rock facility has been in operation since 1983 and is located 60 m below the Baltic Sea floor. Sweden is relying on the geology of its intermediate-depth, crystalline rock cavern repository as the primary migration barrier, although it does use waste form requirements and engineered barriers to augment the geological isolation.

The French l'Aube disposal facility relies heavily on waste form, the engineered concrete vault design, and its liquid collection system to ensure that radionuclides do not reach the biosphere during the institutional control period. If the French design performs as expected, the geology will play no role, through the institutional control period, in preventing radionuclide migration. As already discussed, the long-term performance (post-institutional control) of the facility reverts to primary reliance on the geologic barriers of the facility.

France began disposing of its wastes in 1969 at the La Manche disposal facility in shallow unlined disposal trenches over a layer of gravel in the bottom, backfilled with soil, covered with a plastic sheet, and topped with another layer of soil. Regional monitoring revealed migration of radionuclides from the La Manche site to a nearby water stream. As a result, the facility was redesigned to employ engineered structures and more stringent safety criteria. The La Manche site was chosen largely for reasons of convenience and the geology of the site is not deemed to be ideal for near-surface LLW disposal. The site is now closed and replaced by the l'Aube facility.

Table 6. Disposal facility designs.

Program	Disposal facility	Current disposal method
United Kingdom	Drigg Site	Near-surface concrete vaults (since 1988).
France	Centre de l'Aube	Near-surface concrete vaults.
Sweden	Swedish Final Repository (SFR)	Intermediate-depth crystalline rock cavern.
Canada	Intrusion Resistant Underground Structure (IRUS)	Near-surface concrete vaults.
U.S. DOE	Hanford Low-Level Burial Grounds	Near-surface V-trenches and wide-bottom trenches
	Idaho National Engineering Laboratory	Near-surface pits, trenches, soil vaults.
	Nevada Test Site	
	Area 3	Near-surface disposal in subsidence craters from underground nuclear tests.
	Area 5	Near-surface pits, trenches, boreholes.
	Los Alamos National Laboratory	
	MDA G	Near-surface pits and 20-m deep disposal shafts.
	Oak Ridge Reservation	
	Solid Waste Storage Area 6	Above-grade tumulus.
	Savannah River Site	
	Saltstone	Grout in above-grade vaults (covered with soil, clay, and gravel earthen cap).
	E-Area Vault	Above-grade concrete vaults (covered with soil, clay, and gravel earthen cap).

Britain used trench disposal in a largely clay medium at Drigg until 1988. While maintaining that the risk assessment assured that this method of land disposal was radiologically acceptable, in 1987 BNFL announced a program to improve disposal practices and enhance the visual impact and perception of the Drigg site.[7,17] Trench disposal at the Drigg facility was phased out in preference to more engineered disposal placing containerized, conditioned wastes in concrete vaults.

4.3 Engineered Structures, Surface Barriers, and Waste Form Requirements

Engineered structures, surface barriers, and waste conditioning requirements are increasingly being used by all the disposal programs to help ensure that radionuclide migration is maintained at acceptable levels and to minimize the need for active maintenance of the facility. The decision to impose waste form conditioning requirements and/or engineered structures and surface barriers acknowledges that either greater confinement or additional facility safety associated with redundant migration barriers is desired beyond that afforded by the geology alone.

Tables 7 and 8 briefly summarize the waste conditioning requirements and engineered structures and surface barriers in place at the various disposal facilities included in this study. The table shows that all disposal programs have waste conditioning requirements, although some are more rigorous than others. The waste conditioning requirements for DOE wastes include both generic waste form requirements established in DOE Order 5820.2A and requirements established in the facility waste acceptance criteria, based on site-specific performance assessments. The tables also show that a variety of engineered structures and barriers are used in the programs. Vaults are the most commonly used engineered structure to prevent radionuclide migration, and all facilities plan to employ a cap of some type to limit precipitation infiltration into the wastes.

Volume reduction and physical and chemical stabilization techniques are used to some degree by all the disposal programs. France, Sweden, Canada, and the U.S. DOE use incineration as a treatment option. The UK uses high-force compaction and grouting within the final containers as its primary means of volume reduction and stabilization.[17] Bitumen waste forms are also used extensively in France, Sweden, and Canada, but not commonly by the DOE.

As noted in Section 4.1, all of the programs studied, except the U.S. DOE, currently have only one LLW disposal facility. Thus, the waste conditioning and engineered barrier requirements shown in Table 7 were developed specifically for the single national site, not as a generic national requirement for LLW disposal facilities. As with those of the countries studied, the DOE sites have developed different approaches and designs in accordance with the site-specific geological, hydrological, climatic, and demographic conditions. Some of the DOE sites located in the more arid regions of the United States have determined that the national performance objectives (in DOE Order 5820.2A) for LLW disposal can be met without employing the use of engineered structures. These sites utilize various combinations of reliance on site geology, engineered surface barriers (e.g., earthen cap), and waste form stability requirements. Other DOE sites, in the more humid climates, require waste form stabilization (e.g., grout and high-integrity containers), engineered structures (concrete vaults or tumuli), and engineered surface barriers (such as caps to keep precipitation from the waste).

Table 7. Disposal facility waste conditioning, engineered structure, and surface barrier requirements.

Disposal facility	Waste conditioning	Engineered structures and surface barriers
Drigg Site (UK)	As far as reasonably practicable, waste forms must be insoluble in water and not readily flammable. The UK has two major treatment facilities, the Waste Monitoring and Compaction Facility (WAMAC) and the Drigg Grouting Facility. Volume reduction by high-force compaction is performed at the WAMAC Facility and the compacted containers are filled with grout at the Drigg Facility to fill internal voids.	Concrete vault covered with a water-resistant cap and a soil layer planted with a vegetative cover. Engineered clay base beneath the concrete floor slab.
Centre de l'Aube (France)	Waste must be physically stabilized and radionuclides immobilized for specified concentration thresholds. Waste forms must pass strict tests for physical and chemical stability before the waste form will be accepted for disposal. Generators choose from approved treatment methods including incineration, bitumenation, cementation, polymerized resins, etc. Waste must be in ANDRA-approved containers.	Concrete vault covered with a concrete cap, sealed with a polyurethane and multi-layer cap (clay, bitumen, soil, and a vegetative cover). Space between waste containers is filled with grout or gravel (dependent upon the waste-specific activity). Each vault has a drain system which routes any liquids from the vault to a collection tank. The drain system is located in a concrete tunnel which provides access for inspection and repair.
Swedish Final Repository (SFR)	Each type of waste package must be approved by the Nuclear Power Inspectorate (SKI) and National Institute of Radiation Protection (SSI). Ion-exchange resins are solidified with cement or bitumen. Other processing includes incineration, melting, decontamination, and super-compaction to reduce volumes. Waste must be in solid form; have good chemical, thermal, and mechanical stability; have good immobilization properties; and have a low leach rate. The waste container must be grouted inside steel or concrete drums or boxes.	Crystalline host rock of under-sea caverns, fitted with concrete-walled cells. Each filled cell is backfilled with concrete grout. High-activity waste is disposed in a special cell containing a concrete silo-shaped cavern equipped with internal walls to divide the silo into square shafts. The silo is built on a bed of sand/bentonite (90/10 percent) and the space between the silo wall and the rock is filled with pure bentonite. Once emplaced, waste is surrounded with grout.
Intrusion Resistant Underground Structure (Canada)	Waste is characterized and processed at the Chalk River Waste Treatment Center before disposal. Typical treatment includes incineration, compaction, and solidification. Most drums contain a bitumen waste form produced from liquid-solidification or ash immobilization.	An underground concrete vault with a permeable floor consisting of a 0.3 m thick layer of sand (90%) and clinoptilolite (10%) and a 0.3 m thick layer of sand (90%) and Dochart clay (10%). The clinoptilolite and clay have the capacity to sorb nuclides. The vault is covered by a 1 m thick concrete cap and 1.5 m of sand and soil with a vegetative cover.
U.S. DOE (See Table 8 for specifics of U.S. DOE facilities.)	Requirements are established on a site-specific basis from analytical results of the disposal site performance assessment.	Some sites use near-surface disposal techniques with surface barriers. Other sites use concrete vaults or tumuli in conjunction with surface barriers, if deemed necessary by the disposal site performance assessment.

Table 8. DOE disposal facility waste conditioning, engineered structure, and surface barrier requirements.

Disposal facility	Waste conditioning	Engineered structures and surface barriers
Hanford Low-Level Burial Grounds	No free liquids are accepted and void space must be minimized (generally less than 10 percent of the package volume). All Category 3 (higher activity inventory) waste must be stabilized. Waste may be stabilized by enclosing it in a high-integrity container (HIC), by processing into a stable waste form, or may be shown by analysis to be inherently stable. Processed waste must satisfy performance testing criteria of the NRC Technical Position on Waste Form.	Near-surface disposal in V-trenches and wide-bottom trenches. The waste is backfilled with soil and a final cover, designed to limit the infiltration rate to less than 0.5 cm/yr, is applied to the parts of the disposal facility containing Category 3 (higher activity inventory) wastes.[19]
INEL Radioactive Waste Management Complex	No free liquids are accepted and void space must be minimized. Combustible wastes are incinerated and the ash is stabilized in cement. Large metal shapes are cut down and some wastes are compacted for volume reduction.	Near-surface disposal in pits, trenches, and soil vaults. An earthen cover is placed over the waste during the operational period. Upon closure, a thick soil barrier with a vegetative cover will be emplaced over the operational cover, giving a total soil cover of 5 m.[20]
Nevada Test Site Area 3	No free liquids are accepted and void space must be minimized. Containers must meet stacking strength specifications. Fine particulates must be immobilized. Where practical, waste must be crushed, shredded, and configured to promote waste minimization and to provide a more structurally and chemically stable waste form. Chemical stability must be documented.	Near-surface disposal in subsidence craters from underground nuclear tests. Wastes are disposed using conventional landfill techniques where each layer of waste is covered with 1 m of fill before additional wastes are disposed in the pit.
Area 5		Near-surface disposal in pits, trenches, and boreholes. An earthen cover is placed over the waste during the operational period. Upon closure, a final cap (not yet designed) will be emplaced to enhance facility performance.[21]
Los Alamos National Laboratory MDA G	No free liquids are accepted and void space must be minimized. Fine particulates must be immobilized.	Near-surface disposal in pits and 20 m deep disposal shafts. Waste is placed in the pits and shafts in lifts and crushed tuff is placed in void spaces, between the lifts, and on top of the waste. Filled pits are covered with at least 3 ft. of crushed tuff and 4 inches of top soil and planted with native grasses. Shafts are topped with 1 ft. of concrete shaped to promote drainage away from the shaft.[22]
Oak Ridge Reservation Solid Waste Storage Area 6	No free liquids are accepted and void space must be minimized. Non-compactible wastes are segregated and compactible waste is compacted. The waste form must be stable under the presence of moisture, microbial activity, and internal factors such as radiation effects and chemical changes.	Above-grade tumulus uses concrete rectangular vaults filled with waste, annular spaces are filled with concrete, pre-cast concrete lid is placed on the vault and sealed with bitumen. The vault is subsequently loaded and stacked onto a curbed concrete pad and capped with natural materials. Surface drainage channels divert surface runoff away from the pad.[23]

Table 8. (cont.) DOE disposal facility waste conditioning, engineered structure, and surface barrier requirements.

Disposal facility	Waste conditioning	Engineered structures and surface barriers
DOE Savannah River Site E-Area Vault	No free liquids are accepted and void space must be minimized. Waste packages must not contain greater than 15% void volume. Fine particulates must be immobilized.	Above-grade concrete vaults covered with soil, clay, and a gravel/earthen cap with a vegetative cover. The vaults have a concrete cover to divert surface runoff away from the vaults. The floor of the vault slopes to a drain which runs to a collection sump, which is monitored for radionuclides during the operational period of the facility.[24]
Saltstone	Decontaminated salt solution from the In-Tank Precipitation and Effluent Treatment Facilities is treated by a grouting facility and permanently disposed in above-grade vaults.	Above-grade concrete vaults covered with soil, clay, and a gravel/earthen cap. The saltstone is poured into the vault, leaving approximately 0.3 m from the top of the vault wall to be filled with uncontaminated grout. After all cells are filled, a permanent concrete roof is installed. On closure, soil is placed between the vaults and clay/gravel drainage system with earthen and vegetative cover installed to route precipitation.[25]

5. POST-CLOSURE AND INSTITUTIONAL CONTROL CONSIDERATIONS

The planned institutional control period[b], during which public access to the facility will be controlled, are listed in Table 9 for the programs studied. Sweden is notable in the comparison table since it has no planned institutional control period.

Table 9. Plans for institutional control and use of the facility after institutional control.

Program	Facility name	Institutional control period	Use of facility after institutional controls
U.S. DOE	All DOE sites	Minimum 100 years.	Not specified, TBD on a site-specific basis from technical analysis.
United Kingdom	Drigg Site	Up to 300 years, TBD by regulatory bodies.	Not specified.
France	Centre de l'Aube	300 years.	Unrestricted use.
Sweden	Swedish Final Repository (SFR)	0 years.	Unrestricted use.
Canada	Intrusion Resistant Underground Structure (IRUS)	Not specified. TBD prior to final closure.	Not specified, TBD on a site-specific basis from technical analysis.

The Swedish Final Repository (SFR) facility is located 60 meters beneath the Baltic Sea in crystalline bedrock (gneiss and granite) and is about one kilometer offshore from the harbor at Forsmark. The location has a very low hydraulic gradient and thereby the ground water is almost stagnant. Designers consider that there is no risk of a well being drilled so long as the repository is covered by seawater. Due to the land uplift in Sweden (about 6 mm/yr), the sea bottom above the SFR will become dry land in 1,500 to 2,000 years and the hydraulic considerations will change.[18] This time frame is beyond the typical period of concern for implementing institutional controls.

The SFR facility safety assessment calculates the radiation exposure to the most exposed individual to be 0.0001 mSv/yr (0.01 mrem/yr) during the period the SFR is covered by the Baltic Sea and to be 0.01 mSv/yr (1 mrem/yr) for the period thereafter.[26] Some radiological monitoring of the region will be done but the need for active maintenance based on this monitoring is not anticipated or planned.

France states that it plans to allow unrestricted use of its facilities after the planned 300 year institutional control period. France acknowledges that long-lived radionuclides will still be present upon release of the facility to unrestricted use. Calculations in the l'Aube disposal facility performance assessment include postulated intrusion scenarios during the period when site access is no longer restricted.[27] These scenarios, although called off-normal accident scenarios, include road construction,

b. DOE Order 5820.2A defines "institutional control" as a period of time, assumed to be about 100 years, during which human institutions continue to control waste management facilities.

housing construction, and use of a water well, which are plausible events in an unrestricted access area. The reported external exposure to a housing resident on the site is estimated at 260 mrem/yr from external radiation sources and 23 mrem/yr from inhaled radioactive dust during early years of the free access period. The l'Aube performance assessment uses an allowable exposure limit of 0.005 Sv/yr (500 mrem/yr) to the public during the post-institutional control phase of the facility.[27]

The above allowable dose exceeds the current U.S. DOE guidance given in DOE Order 5400.5, which states that doses from unrestricted use of a disposal facility should not exceed the primary dose limit (100 mrem/yr, except for radon isotopes and progeny). Use of the as low as reasonably achievable (ALARA) process will likely further reduce allowable exposures for DOE properties released for unrestricted use. Proposed regulations (10 CFR 834 and 40 CFR 196) reportedly will deal with these limits, and the allowable exposure limits are expected to be near the 15 mrem/yr range.

The DOE, UK, and Canada have not specified any intended or allowable use of their disposal facilities upon relinquishing institutional control. It has been suggested that restrictions on use of these facilities will be imposed indefinitely, although this would constitute at least a form of administrative institutional control. Allowable future use of the facilities will ultimately be dependent on results of technical analysis in the facility performance assessment that is maintained and updated over time.

The DOE and Canada currently have significant amounts of LLW which contain uranium, thorium, and radium. As discussed in Section 6.2, uranium, thorium, and radium wastes will influence institutional control considerations due primarily to the radon gas, a daughter product of these wastes.

France has imposed isotope-specific concentration limits for Ra-226 and Th-232, which are long-lived (half-life greater than 31 years) alpha-emitters, having radon daughters. The l'Aube limit was established primarily on the results of a pathways analysis during the free access period, which occurs after the 300 year institutional control period.[5] No restrictions related to this concern were found in the Drigg waste acceptance criteria.[28]

Radon emission is not a significant problem for the intermediate-depth Swedish disposal facility since radon gases have a short half-life (Rn-222 has a half-life of 3.8 days) and will decay to lead before escaping the geologic barriers of the intermediate-depth (60 m) facility. The considerations are different in a near-surface disposal facility since the gas can escape the ground and be inhaled by a local inhabitant before it decays.

Canada's regulatory policy statement for radioactive wastes recognizes that it has a significant volume of contaminated equipment and debris originating from uranium mining and milling activities (this is waste other than actual mill tailings) that will require long-term institutional control considerations due to the daughter products of long-lived uranium components.[29] A program separate from the IRUS is being developed to handle the large existing inventory of these "historic wastes." Disposal of these wastes is being planned as a separate effort from currently-produced radioactive wastes and is scheduled to be operational about the year 2000.[7]

Canada is conceptually planning to dispose of newly-generated (non-historic), longer-lived wastes containing radionuclides such as uranium, thorium, carbon-14, and plutonium in some form of rock cavern, possibly in conjunction with a nuclear fuel waste disposal facility, rather than in near-surface disposal facilities. Only a small fraction of the newly-generated waste will not qualify for disposal in a disposal facility like the IRUS. This quantity of Canadian wastes is so small as to not require attention at this time. Instead, Canada plans to store these wastes in engineered facilities for the indefinite future.[6,30]

6. PERFORMANCE ASSESSMENT

6.1 Assessment Performance Objectives

All programs included in this study require technical assessment of the radiological performance of the disposal facility before a proposed LLW disposal operation can be approved. Several different types of performance objectives are used by the various programs as acceptable performance parameters for these assessments. As shown in Table 10, these objectives are expressed as follows:

> Statement of acceptable doses to the general public and to hypothetical inadvertent intruders. (U.S. DOE, Sweden)
>
> Statement of acceptable risk of fatal cancers and genetic defects. (Canada, United Kingdom)
>
> Requirement that fundamental intrinsic safety provisions (waste form and engineered barriers) are present, precluding the possibility of significant release during the facility operational and institutional control periods. After the institutional control period, acceptable radiological exposure to the most exposed member of the public for unrestricted access activities is required. (France)

DOE and Swedish designers base disposal facility radiological performance design considerations largely on calculated exposures of an all-pathways analysis for the most exposed individual. In contrast, Canada and the UK base their radiological performance designs on likelihood of fatalities to the general population (e.g., risk of fatal cancer to public $<10^{-6}$ per year). Approaching facility performance from a perspective of risk to the general population, rather than to the most exposed individual, allows likely events affecting large numbers of people to drive the design instead of a postulated event that might affect and involve relatively few people.

France designs the facility to provide radionuclide migration barriers that confine the nuclides so that essentially no exposure is received by the public during the operational and institutional control periods of the facility. The waste form, engineered barriers, and a liquid collection system are designed to prevent any migration from the disposal vault area, so long as human operators maintain the facility.

It should be noted that the French designers are forced to revert to a different protection scenario at the end of the institutional control period. As faced by all disposal programs, the institutional control period allows significant decay reduction of the radioisotopes with relatively short half-lives, but the longer half-lived isotopes are still a potential exposure radiation hazard that must be addressed. After the 300 year institutional control period, human maintenance and monitoring of the disposal system will cease and reliance can no longer be placed on the integrity of the vault or the liquid collection system. France bases long-term exposure estimates on the conservative assumption that all of the man-made engineered structures and barriers have failed and that the natural geology is the primary mechanism to retard radionuclide migration to members of the public. The allowable exposure limit for this disposal area, returned to unrestricted use, is 0.005 Sv/yr (500 mrem/yr) for the public.

A brief discussion of the Canadian program is provided to illustrate the principles used in a risk-based criteria system. This policy and the basis for the policy is detailed in Canada's Regulatory Policy

Table 10. Time period of compliance and criteria for performance assessment.

Program	Facility name	Time of compliance	Objectives for performance assessment
U.S. DOE	Multiple facilities at DOE sites.	Unspecified by DOE 5820.2A; 10,000 years has been used to-date.	25 mrem/yr to most exposed member of the public. 100/500 mrem/yr (chronic/acute) to inadvertent intruder. Atmospheric releases shall meet the requirements of 40 CFR 61. Groundwater resources shall be protected consistent with Federal, State, and local requirements.
United Kingdom	Drigg Site	Unspecified.	Risk of fatal cancer to public $<10^{-6}$ per year.
France	Centre de l'Aube	Unspecified. PA calculations exceed 10^6 years.	Designed so that it possesses intrinsic safety (effectively no release to the environment) through the institutional control period, based on the reliability of its first two systems of containment. Reliance is placed on geological barriers after the institutional control period with 500 mrem/yr as the maximum allowable exposure to the public.
Sweden	Swedish Final Repository (SFR)	10,000 years	0.1 mSv/yr (10 mrem/year) to public.
Canada	Intrusion Resistant Underground Structure (IRUS)	10,000 years	Risk of fatal cancers and serious genetic effects $<10^{-6}$ per year.

Statement on disposal of radioactive wastes.[29] In this policy, Canada establishes the following general requirement:

> "The predicted radiological risk to individuals from a waste disposal facility shall not exceed 10^{-6} fatal cancers and serious genetic effects in a year, calculated without taking advantage of long-term institutional controls as a safety feature."

The Canadian policy document explains that this level of risk, 1×10^{-6} in a year, was chosen because of other activities that consider this level of individual risk to be insignificant in daily lives. Since the probability of fatal cancers and serious genetic effects is approximately 2×10^{-2} per Sv (2×10^{-4} per rem), the probability of these health effects associated with a dose of 1 mSv (100 mrem) is 2×10^{-5}. (These reported health effect values are from the ICRP Publication 26 and are a factor of approximately 2 less than more recent values given in ICRP Publication 60). To put this in perspective, a risk of 10^{-6} (1 in a million) is the risk associated with a dose of 0.05 mSv (5 mrem) in a year. Where it is reasonable to assume that the probability of the scenario approximates unity, the risk is simply the product of dose and the probability of the health effect per unit dose. On the other hand, if an event is highly unlikely, such as an inadvertent intrusion into an undesirable location, the probability of the event will allow designers to discount the unlikely event and to use a more probable scenario as the basis for designing the facility.

Canada also provides guidance for the case where the above risk criteria cannot be met. If there is no practicable method of fully meeting the above health risk, an optimization study must be performed to determine the preferred option. A disposal facility built under these circumstances shall be: 1) compatible with the results of such a study and 2) such that the predicted risk to individuals does not exceed that which is presently accepted from current operations involving the same wastes.

The approach taken by DOE is briefly discussed to illustrate the principles used in a dose-based criteria system. DOE established the following criteria as the basis for determining acceptable performance of its LLW disposal facilities:

(1) The effective dose equivalent to any member of the public shall not exceed 25 mrem/yr.

(2) Releases to the atmosphere shall meet the requirements of 40 CFR 61.

(3) The committed effective dose equivalent received by individuals who inadvertently intrude into the facility after the loss of active institutional control (100 years) will not exceed 100 mrem/yr for continuous exposure or 500 mrem for a single acute exposure.

(4) The groundwater resources shall be protected consistent with federal, state, and local requirements.

Designers must postulate and analyze potential credible scenarios that may impact the above criteria. In practice, the likelihood of the events are normally determined by an analyst's judgement rather than formal numerical calculation. It is possible for relatively unlikely events or events that affect relatively few people to drive the facility design.

6.2 Time of Compliance

The time of compliance is defined as a specific time period during which the performance of a LLW disposal facility or the disposal system (disposal facility and environmental conditions) must be shown to satisfy the performance objectives. Table 10 (above) shows the time period of compliance for each of the programs. All the disposal programs recognize that the wastes contain at least a small amount of long-lived radioactive isotopes requiring long-term consideration. Canada and Sweden both have a time period of compliance of 10,000 years. France and the UK do not specify a specific time of compliance, although they are analyzing the long-term performance of their facilities.

The performance objectives in DOE Order 5820.2A do not specify a time period over which they are to be applied. Facility-specific performance assessments written to date have used 10,000 years and recognized the time of peak dose. Times from 1,000 to 10,000 years have been proposed as an official time of compliance but no formal policy has been approved.[31]

As discussed in Section 3, France, Sweden, and the UK limit the concentration of alpha-emitting isotopes in their LLW wastes to approximately 100 nCi/g. DOE imposes the limit that only transuranium alpha-emitting isotopes (with half-lives greater than 20 years) must be at concentrations less than 100 nCi/g, although Field Offices are allowed to specify other alpha-contaminated wastes that must be managed as transuranic waste. This allows the DOE to dispose of wastes having >100 nCi/g uranium, thorium, and other non-transuranium alpha-emitting isotopes (shown in Table 4), assuming that the performance assessment shows acceptable exposures over the time period of compliance. Most DOE sites limit inventories of these wastes through their facility waste acceptance criteria.

The time period of compliance used in the performance assessment is significant when considering the impacts of uranium isotopes that require significant amounts of time to reach equilibrium with their radioactive daughter products. The time required for radium and its decay products (principally Rn-222) to reach equilibrium with initially pure uranium is approximately 10^5 years for U-238 plus U-234 and 10^6 years for U-238 only.[31] Therefore, the radiological exposures from uranium daughters are increasing with time, for hundreds of thousands of years. If a disposal site uses a time period of compliance of 10,000 years or less, the impact of uranium and other long-lived alpha-emitters on exposures to distant-future inhabitants is still increasing at the end of the time period of compliance.

Some believe that this approach is reasonable since projections of impacts that far into the future are too uncertain to be considered realistic or useful for making decisions. Many factors such as proximity to inhabitants and migration and solubility of the wastes will affect the magnitude of the future exposure from the daughter products of long-lived isotopes.

Canada and DOE, in particular, have considerable quantities of these uranium, radium, and thorium wastes associated with their early uranium enrichment and defense activities. Canada has not yet decided what disposal option will be used for its large quantities of historic wastes. Potential solutions include prescribing a flux limit for discharge of radon gas, implementing very long-term institutional control of the disposal facility, and/or burying the waste deeper to prevent the radon gas from escaping the geology before decaying.

7. SUMMARY

This study revealed that notable differences exist between the programs of the U.S. DOE and the other countries concerning how they dispose of LLW. Table 11 briefly summarizes many of the differing practices of the disposal programs included in this study. The reasons for these differences are complex and dependent on many variables both technical and political.

Even though each disposal program studied utilizes a different LLW classification system, it is interesting to note that all programs recognize the need to limit long-lived radionuclides in the disposal facility. The UK, France, and Sweden all limit alpha-emitting radionuclide concentrations to approximately 100 nCi/g. DOE Order 5820.2A limits alpha-emitting transuranium radionuclides (having half-lives greater than 20 years) to concentrations less than 100 nCi/g. The lighter alpha-emitter elements (including uranium, thorium, and radium) are thus allowable unless restricted by a site-specific waste acceptance criteria based on the results of the facility performance assessment. Canada acknowledges that the waste must have a hazardous lifetime of less than 500 years but has not yet put generic quantitative values on the alpha-emitting isotopes.

Various criteria are used to limit allowable beta-gamma activity in waste classified as LLW. The DOE, France and Canada impose classification limits, related to beta-gamma emitting radionuclides, by exclusion. That is, if a waste is radioactive, but it is not high-level waste, transuranic waste or greater-than-Class C (applicable to the U.S. DOE only), uranium mill tailings, or spent fuel, then it is classified as LLW. The UK specifies quantitative activity concentration limits for beta-gamma activity in its LLW classification scheme. Sweden uses the IAEA system, which defines a maximum heat generation criteria (2 kW/m^3) for the waste, which is related to the beta-gamma activity. Waste disposal facilities sometimes further limit both alpha and beta-gamma emitters, on an isotope specific basis, in their waste acceptance criteria based on the analytical results of the site-specific performance assessment.

Significant differences were found when investigating how facilities implement their limits. The French use calculated alpha-emitter concentrations 300 years after time of acceptance, allow up to 500 nCi/g per container (on a limited exception basis), and include the mass of the waste container and any solidification agent when calculating alpha concentrations. In contrast, the U.S. DOE uses alpha concentrations at the time of waste form assay, rather than a projected inventory at some date in the future. DOE does not allow exceptions to their 100 nCi/g limit and uses only the mass of the waste matrix (excluding the disposal container) to calculate alpha concentrations, for purposes of determining if the waste will be classified as LLW or transuranic waste. These practices can make appreciable difference in allowable waste forms that may be called LLW.

Sweden, the UK, and Canada have established accepted definitions of "exempt" or "very low-level" waste which can be disposed in non-licensed facilities with minimal or no disposal constraints. Sweden has adopted the IAEA recommended classification system that establishes acceptable dose consequences to the general public resulting from exempt waste at less than 0.01 mSv/yr (1 mrem/yr). The IAEA recently published suggested radionuclide-specific clearance levels for comment by IAEA member countries.[4]

All programs studied require waste conditioning before land disposal in near-surface disposal facilities. These conditioning requirements vary but universally no sites accept free liquids, and hazardous components must typically be eliminated or stabilized. Waste conditioning requirements vary from one disposal site to another based on the site-specific needs established in the facility performance assessments and on national policy requirements.

Table 11. Practices of national programs for disposal of low-level waste.

Topic	U.S. DOE	France	Sweden	United Kingdom	Canada
Working Definition of Exempt Waste (i.e., below regulatory concern) Is In Use			X	X	X
Waste Conditioning Performed	X	X	X	X	X
Disposal Practices					
Near-surface disposal utilizing an engineered structure (e.g., vaults, tumuli)	X	X		X	X
Near-surface disposal without an engineered structure (e.g., vaults, tumuli)	X				
Intermediate-depth geologic disposal (60 meters below the Baltic Sea floor)			X		
Cap Design					
Solid Concrete	X (Oak Ridge and Savannah River)	X	NA		X
Earthen Cover	X	X	NA	X	X
Active Institutional Controls Planned After Closure	X	X		X	X
Time Period of Compliance	Unspecified 10,000 years is precedence	Unspecified	10,000 years	Unspecified	10,000 years
Performance Criteria					
Based on most exposed individual	X	X (free access period)	X		
Risk-based chance of cancer				X	X

X Indicates that the row topic is applicable to the indicated national program.

All programs studied that conduct near-surface LLW disposal use engineered structures (such as vaults) for LLW disposal. DOE facilities located in the more humid areas of the United States (Savannah River and Oak Ridge) use a similar system. The more arid DOE disposal sites (Nevada, Los Alamos, Hanford, and Idaho) do not have engineered structures in their facility design, but do use surface barriers such as water-repellent layers (clay, asphalt, concrete), capillary barriers (hydraulic breaks), and rock layers (riprap, gravel) to minimize water infiltration.

The variety of combinations of engineered structures and barriers found at the various LLW disposal facilities studied is an indication that the technical performance of each near-surface disposal site is based on many factors including geology, climate, and hydrology. An analysis (performance assessment) of the disposal system is a universally accepted method of judging the technical adequacy of a disposal system.

Demographic, economic, socio-political, and institutional factors also play a significant role in defining acceptable disposal solutions. Design conservatism is often reflected in disposal facility designs to gain public acceptance of the facility.

Unlike the other disposal programs, Sweden has elected to dispose of their LLW in an intermediate-depth disposal facility, located in crystalline rock, approximately 60 meters beneath the Baltic Sea floor. Sweden uses intermediate-depth disposal primarily due to the requirement that active institutional controls and corrective actions after closure of the facility should be minimized or eliminated, if possible.

With the exception of Sweden, all of the countries and DOE utilize some form of engineered earthen covers to shed water and protect waste from the direct weather elements. Concrete caps are utilized in France and Canada, and at the humid DOE sites (Oak Ridge and Savannah River). Sweden has chosen an alternative design approach of intermediate-depth disposal in crystalline rock, 60 m below the Baltic Sea floor. The UK vault design does not use a concrete cap beneath its multiple-layered earthen cover.

Institutional control periods are planned for all the disposal programs studied, with the exception of Sweden, which plans no active institutional control measures. The undersea location of the Swedish disposal facility eliminates the need for physical institutional control measures after the entrance is sealed.

All of the disposal programs studied recognize the need for analysis of long-term periods of performance for waste analysis efforts. Sweden and Canada use 10,000 years as the time period of compliance. DOE, France and the UK do not specify the time period of compliance for their facilities. Existing DOE facility-specific performance assessments written to date have used 10,000 years and generally determine the time of peak dose.

Several approaches to defining criteria for assessment of acceptable disposal facility performance are utilized by the various nations. The U.S. DOE and Sweden specify criteria for the most exposed individual as the criteria to establish performance and design requirements. The UK and Canada have adopted risk-based criteria based upon the likelihood of cancer or genetic defects. France requires features intended to eliminate any significant radionuclide migration from the vault area during the operational and institutional control periods. France places no reliance on the geological barriers until after the institutional control period. At that point, the facility changes to performance criteria of allowable exposure to the most exposed individual.

The various programs studied have established different approaches to LLW disposal. These include near-surface disposal with engineered structures and/or surface barriers, intermediate-depth geologic disposal, and deep geologic disposal. All programs studied utilize varying degrees of a multi-barrier approach to isolating radionuclides from the environment and all implement engineered barriers and/or waste conditioning requirements beyond those of traditional shallow land disposal in trenches. The policies of each nation are influenced by geographic, climatic, demographic, economic, socio-political, and institutional factors.

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